

May 25, 2001

Mr. Harold W. Keiser
Chief Nuclear Officer & President
PSEG Nuclear LLC - X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF
AMENDMENT RE: INCREASE LICENSED POWER LEVELS FROM 3,411 MWT
TO 3,459 MWT (TAC NOS. MB0521 AND MB0522)

Dear Mr. Keiser:

The Commission has issued the enclosed Amendment Nos. 243 and 224 to Facility Operating License (FOL) Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Salem FOLs and Technical Specifications (TSs) in response to your application dated November 10, 2000, as supplemented by letters dated December 5, 2000, March 28 and April 2, 2001, and three letters dated April 20, 2001 (LRN-01-0099, LRN-01-0115, and LRN-01-0123).

These amendments increase the licensed power level by approximately 1.4% from 3,411 megawatts thermal (MWt) to 3,459 MWt. The changes are anticipated to increase each unit's net electrical output by 16 MWe. The request is based on the installation of the CE Nuclear Power LLC Crossflow ultrasonic flow measurement system with its ability to achieve increased accuracy in measuring steam generator feedwater flow.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Robert J. Fretz, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures: 1. Amendment No. 243 to
License No. DPR-70
2. Amendment No. 224 to
License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

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ACCESSION NUMBER: ML011350051

TEMPLATE=NRR-058

* SE input provided. No major changes made.

OFFICE	PDI-2/PM	PDI-2/LA	SRXB/SC*	SPLB/SC*	SPSB/BC*	EMCB/SC*	EMEB/SC*	EEIB/SC*
NAME	RFretz	TLClark	FAkstulewicz	GHubbard	PWilson	KWichman	KManoly	CHolden
DATE	5/24/01	5/25/01	01/16, 04/26 and 04/30/01	04/17/01	04/04/01	04/18 and 04/30/01	05/22/01	05/08/01

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NAME	EMarinos	DTrimble	RWeisman	EAdensam for JClifford	EAdensam	CCarpenter for JZwolinski	TCollins for BSheron	DMatthews for SCollins
DATE	05/09/01	04/22/01	5/24/01	5/25/01	5/25/01	5/25/01	5/25/01	5/25/01

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PSEG NUCLEAR LLC
EXELON GENERATION COMPANY, LLC
ATLANTIC CITY ELECTRIC COMPANY
DOCKET NO. 50-272
SALEM NUCLEAR GENERATING STATION, UNIT NO. 1
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 243
License No. DPR-70

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:
 - A. The application for amendment filed by the PSEG Nuclear LLC, Exelon Generation Company, LLC, and Atlantic City Electric Company (the licensees) dated November 10, 2000, as supplemented by letters dated December 5, 2000, March 28 and April 2, 2001, and three letters dated April 20, 2001 (LRN-01-0099, LRN-01-0115, and LRN-01-0123), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Operating License and the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-70 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 243, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA DMatthews for/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License and
Technical Specifications

Date of Issuance: May 25, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 243

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following pages of the Facility Operating License and Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages

License, page 4

1-5

2-2

2-8

3/4 4-26

3/4 4-27

3/4 7-2

3/4 7-3

6-24a

B 3/4 4-6

B 3/4 4-7

B 3/4 4-8

B 3/4 4-9

B 3/4 4-10

B 3/4 4-11

B 3/4 4-12

B 3/4 4-15

B 3/4 7-1

Insert Pages

License, page 4

1-5

2-2

2-8

3/4 4-26

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3/4 7-3

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B 3/4 4-6

B 3/4 4-7

B 3/4 4-8

B 3/4 4-9

B 3/4 4-10

B 3/4 4-11

B 3/4 4-12

B 3/4 4-15

B 3/4 7-1

PSEG NUCLEAR LLC
EXELON GENERATION COMPANY, LLC
ATLANTIC CITY ELECTRIC COMPANY
DOCKET NO. 50-311
SALEM NUCLEAR GENERATING STATION, UNIT NO. 2
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 224
License No. DPR-75

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:
 - A. The application for amendment filed by the PSEG Nuclear LLC, Exelon Generation Company, LLC, and Atlantic City Electric Company (the licensees) dated November 10, 2000, as supplemented by letters dated December 5, 2000, March 28 and April 2, 2001, and three letters dated April 20, 2001 (LRN-01-0099, LRN-01-0115, and LRN-01-0123), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Operating License and the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 224, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA DMatthews for/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License and
Technical Specifications

Date of Issuance: May 25, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 224

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Replace the following pages of the Facility Operating License and Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages

License, page 3

1-5

2-2

2-8

3/4 4-28

3/4 4-29

3/4 7-2

6-24a

B 3/4 4-7

B 3/4 4-8

B 3/4 4-9

B 3/4 4-10

B 3/4 4-11

B 3/4 4-12

B 3/4 4-13

B 3/4 4-16

B 3/4 7-1

Insert Pages

License, page 3

1-5

2-2

2-8

3/4 4-28

3/4 4-29

3/4 7-2

6-24a

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B 3/4 4-9

B 3/4 4-10

B 3/4 4-11

B 3/4 4-12

B 3/4 4-13

B 3/4 4-16

B 3/4 7-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NOS. 243 AND 224 TO FACILITY OPERATING
LICENSE NOS. DPR-70 AND DPR-75
PSEG NUCLEAR LLC
EXELON GENERATION COMPANY, LLC
ATLANTIC CITY ELECTRIC COMPANY
SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2
DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated November 10, 2000, as supplemented by letters dated December 5, 2000, March 28 and April 2, 2001, and three letters dated April 20, 2001 (LRN-01-0099, LRN-01-0115, and LRN-01-0123), PSEG Nuclear LLC (PSEG) submitted a request for changes to the Salem Nuclear Generating Station, Unit Nos. 1 and 2 (Salem), Facility Operating Licenses (FOL) and Technical Specifications (TSs). The requested changes would increase the licensed power level for each unit by approximately 1.4% from 3,411 megawatts thermal (MWt) to 3,459 MWt. The request is based on the installation of the CE Nuclear Power LLC (CENP) Crossflow ultrasonic flow measurement (UFM) system with its ability to achieve increased accuracy in measuring feedwater flow. CENP topical report CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," documents the theory, design, and operation of the Crossflow system.

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specific core thermal power level. The power level is indicated in the control room by neutron flux instrumentation that is calibrated to correspond to core thermal power. Core thermal power is validated by a nuclear steam supply system (NSSS) energy balance calculation. The accuracy of this calculation depends primarily upon the accuracy of feedwater flow, temperature, and pressure measurements.

The thermal power levels assumed in a plant's design basis transient and accident analyses must bound the potential range of power levels at which the plant could be operated. The uncertainty of calculating values of core thermal power is factored into the allowable thermal power levels to reduce the likelihood of exceeding the power levels assumed in the analyses. At one time, Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix K, required licensees to base their transient and accident analyses on an assumed power level of

at least 102% of the licensed thermal power level. This was to allow for uncertainties in determining thermal power (e.g., instrument measurement uncertainties). The 2% power margin uncertainty value was intended to address uncertainties related to heat sources in addition to instrument measurement uncertainties. The U.S. Nuclear Regulatory Commission (NRC) concluded, at the time of the original ECCS rulemaking, that the 2% power margin requirement was based solely on considerations associated with power measurement uncertainty as is reflected in Appendix K.

Appendix K to 10 CFR Part 50 did not require demonstration of the power measurement uncertainty and mandated a 2% margin, notwithstanding that the instruments used to calibrate the neutron flux instrumentation may be more accurate than originally assumed in the ECCS rulemaking. On June 1, 2000, the NRC published a final rule (65 FR 34913) that allows licensees to justify a smaller margin for power measurement uncertainty by using more accurate instrumentation to calculate the reactor thermal power and thereby calibrate the neutron flux instrumentation.

In its application, PSEG requested approval to increase Salem's licensed thermal power levels based on the installation of the CENP Crossflow measurement system. The Crossflow system is designed to improve the accuracy of feedwater flow rate measurement, which is used, in part, to calculate reactor thermal power. The improved flow measurement instrumentation would allow PSEG to operate Salem with a reduced margin between the actual power level and the 102% margin previously used in the licensing basis ECCS analyses.

3.0 EVALUATION

The NRC staff's review of PSEG's 1.4% power uprate license amendment application contains the following subsections:

- 3.1 Nuclear Steam Supply System (NSSS) Evaluation
- 3.2 NSSS/Balance-of-Plant Interface Systems Evaluation
- 3.3 Balance-of-Plant Systems Evaluation
- 3.4 Instrument and Controls Systems
- 3.5 Electrical Systems
- 3.6 Nuclear Steam Supply System Accident Evaluation
- 3.7 Radiological Analysis of Design Basis Accidents
- 3.8 TS Changes
- 3.9 Human Factors

3.1 Nuclear Steam Supply System (NSSS) Evaluation

3.1.1 Reactor Pressure Vessel (RPV)

3.1.1.1 RPV Structural Evaluation

PSEG evaluated the effect of changes in several operating parameters that result from the increased power level on the RPV stress range and fatigue usage factors (CUFs). PSEG stated that the power increase will result in the revised design parameters given in Tables 2-1 and 2-2 of Attachment 1 to its November 10, 2000, application. These parameters include the limiting hot-leg temperature (T_{hot}) and cold-leg temperature (T_{cold}), and the NSSS design

transients. As a result of the proposed power uprate, T_{hot} will increase by 0.5 F° and T_{cold} will decrease by 0.5 F°. PSEG concluded that since reactor pressure will remain unchanged, T_{hot} and T_{cold} at the uprated power level are still bounded by the design basis temperatures in the existing reactor vessel stress report. Therefore, the stress intensities and CUFs remain within the applicable limits of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code). The staff finds this acceptable because the stress intensities and CUFs will remain within Code-allowable limits.

Additionally, PSEG evaluated the primary-water-induced stress corrosion cracking susceptibility for Salem, Unit Nos. 1 and 2, and stated that the increase is negligible. The staff concurs with PSEG's conclusions and finds this acceptable.

3.1.1.2 Reactor Pressure Vessel (RPV) Integrity - Neutron Irradiation

Neutron irradiation affects the ductility of the RPV. The Commission's requirements for acceptable RPV fracture toughness properties are contained in 10 CFR Part 50, Appendix G, and 10 CFR 50.61, "Fracture Toughness Requirements for Protecting Against Pressurized Thermal Shock [(PTS)]". To satisfy both requirements, PSEG performed a three-dimensional evaluation of the effect of the power uprate on the neutron and gamma-ray fields. The results (increased neutron and γ -fluence) are incorporated in the proposed pressure-temperature (P-T) curves and the new RT_{PTS} limits (the reference temperature at the nil ductility transition, RT_{NDT} , for the material at end of life) defined in 10 CFR 50.61 to 32 Effective Full Power Years (EFPY) of operation. The use of ASME Code Case N-640 was incorporated into the calculation of the P-T limits and Appendix G flange requirements. Use of this code case was approved by an exemption to NRC regulations dated May 25, 2001. The NRC staff's evaluation of issues associated with RPV integrity and neutron irradiation concerns are presented in Section 3.8.2 of this safety evaluation.

3.1.1.3 Reactor Internals

Reactor internals support the fuel, control elements and instrumentation, and transmit static and dynamic loads to the pressure vessel. This subsection considers the capability of the reactor internals to perform their function at the uprated power level.

Reactor Core Support Structures

By letter dated April 20, 2001, PSEG provided additional information requested by the NRC staff with regard to its evaluation of the reactor vessel core support and internal structures. The limiting reactor internal components evaluated include the lower core plate, core barrel, baffle plate, baffle/barrel region bolts, and the upper core plate. PSEG indicated that, since the reactor internal components were designed prior to the introduction of Subsection NG of the ASME Code, Section III, its evaluation used the original Westinghouse design criteria, as documented in the Salem FSAR, which are similar to the criteria described in Subsection NG of the Code.

PSEG evaluated critical reactor internal components using the revised design conditions provided in Tables 2-1 and 2-2 of Attachment 1 to its November 10, 2000, application. PSEG indicated that the 1.4% uprate does not change the current design basis seismic and loss of

coolant accident (LOCA) loads. For the baffle-barrel region and the upper core plate, PSEG stated that the current structural and thermal analyses of record for Salem remain bounding for the power uprate condition. In its April 20, 2001, letter, PSEG provided the maximum calculated stress intensity and CUFs for the lower core plates. The maximum stress intensity and CUF are less than the acceptable limits. The remaining reactor internal components are less limiting. In addition, PSEG indicated that the current analysis of record for the flow induced vibration is bounding for the power uprate. As a result of these evaluations, PSEG concluded that the reactor internal components at Salem will be structurally adequate for the proposed power uprate conditions. The staff concurs with PSEG's conclusion.

Furthermore, PSEG indicated that the current analysis of record for flow induced vibration is bounding for the power uprate. As a result of its evaluation, PSEG concluded that the reactor internal components at Salem will be structurally adequate for the proposed power uprate conditions. The staff concurs with PSEG's assessment.

Thermal-Hydraulic Systems Evaluation

PSEG estimated the core bypass flow to be somewhat lower but will remain adequate to provide cooling to internal components and the reactor pressure vessel (RPV) head. PSEG evaluated the reactor internal components' hydraulic lift forces, and concluded that they are enveloped by the analysis of record. The staff concludes, for the same reason, that no further reexamination is warranted.

PSEG reexamined the baffle jetting in view of the power uprate. The examination showed that the bypass leakage momentum flux, which is responsible for fuel rod oscillations, did not change as a result of the power uprate. TS 3.1.3.3 requires that the control rod cluster assembly maintains drop time of less than 2.7 seconds. PSEG stated that an evaluation confirmed that the drop time will continue to be within the limits of the TS. The staff finds this acceptable because the rod drop times will remain within TS limits.

Mechanical Evaluation

PSEG reexamined the current LOCA design basis, with respect to the structural effects of the seismic operating basis earthquake and the safe shutdown earthquake loads. PSEG determined that the changes due to the proposed power uprate are enveloped by the current analysis of record. Likewise, the flow- and pump-induced vibrations due to the hot-leg temperature (T_{hot}) and cold-leg temperature (T_{cold}) changes are also enveloped by the analysis of record. Therefore, the staff concludes that this is acceptable and that no further examination is warranted.

Structural Evaluation

Potential changes in the structural integrity of the reactor internals could result from changes in the thermal gradients and the increased gamma heat deposition to reactor internal components. PSEG examined the baffle-to-baffle and baffle-to-former bolts in the core baffle region for changes to their deadweight, pressure differential, seismic, preloads and thermal loads resulting from reactor coolant system (RCS) and gamma heating. A structural

assessment indicated that the current analysis of record is valid. Indeed, the gamma heating deposition rates will be lower due to low leakage loadings which have been adopted for longer fuel cycles since the original analysis was performed.

The lower support plate supports the lower core plate and is subjected to significant gamma heating due to its proximity to the core. PSEG conducted a structural evaluation that included the parameters of the power uprate which demonstrated that the structural integrity of the lower core plate is not adversely affected and the calculated value of the fatigue utilization factor will remain lower than 1.0.

Therefore, the staff finds the licensee's evaluation of the impact of the 1.4% power uprate on reactor internals to be acceptable because the structural integrity of the lower core plate will be maintained and the calculated value of the fatigue utilization factor will remain lower than 1.0. In addition, the results will continue to be bounded by Salem's analysis of record.

3.1.1.4 Control Rod Drive Mechanisms (CRDMs) and CRDM Nozzles

The upper head of the control rod drive mechanism (CRDM) is exposed to hot leg fluid temperatures. The maximum T_{hot} used in the design basis was 616.3 °F. The maximum T_{hot} estimated for the proposed power uprate is 613.1 °F. Therefore, the power uprate conditions will be bounded by the analysis of record.

3.1.1.5 Nuclear Fuels

Fuel Assemblies

The staff reviewed issues associated with the impact during seismic and LOCA events that the 1.4% increase in nominal power will have on the structural integrity of the three types of fuel assemblies found in the Salem Unit Nos. 1 and 2 reactor cores. Since the core plate motions are not affected by the 1.4% power increase, the homogeneous and mixed-core seismic analyses are still valid. Similarly, Salem's LOCA analyses were performed at 1.02 times licensed power, and are, therefore, still valid.

Nuclear Fuels

The proposed power uprate will impact the peaking factors, rod worths, reactivity coefficients, shutdown margin, and the kinetics parameters. All of these quantities are subject to core design control. The methods involved in fuel cycle design remain unchanged because the 1.4% power uprate is within the parameter range of the current methodology.

The thermal hydraulic design was performed at the uprated power of 3,459 MWt. VANTAGE 5H fuel assemblies with and without intermediate flow mixers (IFM) were assumed in this design, and a reduced radial factor ($F_{\Delta H}$) was credited for burnt assemblies without intermediate flow mixers. The design method used to calculate departure from nucleate boiling (DNB) is the Westinghouse revised thermal design procedure (RTDP). The WRB-2 (Westinghouse Round Rod Bundle) correlation was used for 17x17 fuel assemblies with IFMs and the WRB-1 for fuel assemblies without IFMs. New DNBR (DNB Ratio) limits, axial offset, and accident

acceptability limits were calculated to support reactor operation at the uprated power level. The analysis showed that the DNB design basis is met and the 1.4% uprate is within the bounds of the Final Safety Analysis Report (FSAR).

The fuel rod design was evaluated for an equilibrium cycle reload at the 3,459 MWt power level. Burnups up to 62,000 MWD/MTU for three cycle loadings were considered. Integral burnable absorbers were also assumed in a range of values. The results of these evaluations indicated that the fuel meets all of the fuel rod design limits.

3.1.1.6 Reactor Vessel Evaluation Conclusions

Based on its review, the NRC staff finds that the RPV for Salem, Unit Nos. 1 and 2, will continue to ensure that specified acceptable fuel design limits set forth in the Salem FSAR are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences at the uprated power level of 3,459 MWt. Therefore, the staff finds this acceptable.

3.1.2 Reactor Coolant System (RCS)

The RCS consists of four heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump that circulates the water through the loop, the reactor vessel, and a steam generator, where heat is transferred to the main steam system. The RCS contains a pressurizer that controls the RCS pressure through electrical heaters, water sprays, power-operated relief valves (PORVs) and spring-loaded safety/relief valves. The steam discharged from the PORVs and safety/relief valves flows through interconnecting piping to the pressurizer relief tank.

PSEG performed various assessments and demonstrated that the RCS design basis functions would be met at the revised design conditions. The minimum required spray flow of 800 gpm can be achieved for 1.4% uprate conditions. The maximum expected T_{hot} (613.1 °F) at the revised conditions is well below the RCS loop design temperature of 650 °F. Therefore, all calculations performed using the RCS loop design temperature remain bounding. The nominal full-load pressurizer steam volume is unaffected by the uprate since the maximum RCS average temperature of 577.9 °F has not changed. Therefore, the existing PORV and safety/relief valve discharge analysis is unaffected.

3.1.2.1 Reactor Coolant Piping and Supports

Reactor Coolant Loop Piping, Equipment, and Branch Nozzles

As discussed in PSEG's application dated November 10, 2000, Attachment 1, Section 6.5, the power uprated conditions do not change the hydraulic forcing functions; therefore, the loads to the reactor coolant loop piping and nozzles are bounded by the existing analysis. The conditions in the pressurizer surge line are also bounded by the analysis of record. Indeed, the new operating conditions will be conservative because the increased T_{hot} (by 0.5 F°) reduces the surge line stratification ΔT . In addition, a zero gap between the equipment and the support structure is used. The power uprate will increase T_{hot} by only 0.5°F, for which the dimensional changes are well within the fabrication tolerance limits. Therefore, the existing equipment support analysis remains bounding.

Leak Before Break

The NRC staff has reviewed the information submitted by PSEG regarding the potential impact of the proposed Salem power uprate on the acceptability of the leak-before-break (LBB) status for Salem's RCS piping. LBB for Salem was justified in Westinghouse Topical Report WCAP-13659, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Salem Generating Station Units 1 and 2," by demonstrating that: (1) there exists margin between the critical crack size (i.e., a conservatively assumed flaw that would result in a pipe rupture) and the postulated crack that yields a detectable leak rate; (2) there is sufficient margin between the leakage through the postulated crack and the leak detection capability; (3) there exists margin on the applied load; and (4) the fatigue crack growth potential is negligible.

In Section 5.6 of PSEG's submittal, it was shown that there is no impact on the RCS piping loads. The impact on piping loads due to the 0.5 F° change in the coolant temperature will also be negligible, thus, the NRC staff concludes that the LBB analysis of record remains bounding. Therefore, the staff has concluded that, in accordance with the provisions of 10 CFR Part 50, Appendix A, GDC 4, the dynamic effects from postulated breaks of Salem's RCS piping may continue to be excluded from the licensing basis of the facility for post-power uprate conditions.

Reactor Trip and Engineered Safety Feature Actuation System Setpoints

PSEG evaluated steam generator low-low and high-high water reactor trip functions, and found that the impact of the power uprate was negligible. Therefore, the staff concurs with PSEG's conclusion that no changes to Reactor Trip and Engineered Safety Feature Actuation System (ESFAS) setpoints are necessary.

Reactor Vessel, Loop, and Steam Generator Tube Degradation Mechanisms

The proposed rated power increase of 1.4% with an uncertainty of 0.6% does not affect the magnitude of the heat sources mandated by 10 CFR Part 50, Appendix K for LOCA analyses. Therefore, the staff concludes that the existing LOCA hydraulic forces analyses of record supporting Salem Unit Nos. 1 and 2 are acceptable.

Revised Thermal Design Procedure Uncertainty

The revised thermal design procedure, as described in Westinghouse Topical Report WCAP-1351, takes the uncertainties in thermal design parameters systematically into account. The proposed increase in the power level, with an associated smaller uncertainty, will not increase the uncertainties of parameters computed with the revised thermal design procedure.

3.1.2.2 Pressurizer

PSEG's review of the changes in the RCS hot-leg (T_{hot}) and cold-leg (T_{cold}) temperatures, due to the proposed 1.4% increase in the power level, shows very small changes. These changes are within the envelope of the current stress analyses. In addition, since the design transients are unaffected by the proposed increase in power, there is no impact on the current stress and fatigue analyses.

3.1.2.3 Pressurizer Overpressure Protection System (POPS)

The changes to the full-power operating parameters do not affect the POPS, and therefore, the POPS analysis is unaffected. PSEG reviewed the revised P-T curve pressure limits against the POPS analysis results and verified that they were bounded by the criteria in ASME Code Case N-640.

3.1.2.4 Reactor Coolant Pumps and Motors

The analysis of record is based upon a 544.8 °F steam generator temperature. The corresponding steam generator (SG) temperature with the power uprate is 542.5 °F, which remains within the bounding analysis. The current continuous reactor coolant pump motor operation analysis is based on a SG outlet temperature of 528.7 °F at a flow of 89,200 gallons per minute (GPM) per loop. The lowest steam generator RCS outlet temperature in the proposed power uprate conditions is 530.0 °F at a flow of 89,700 gpm/loop. The lower RCS water density at 530.0 °F will result in a slightly reduced MWe load for the RCP motors at continuous hot power operation. Therefore, the staff concludes that the analysis of record bounds the motor operation at the uprated conditions.

3.1.2.5 RCS Evaluation Conclusion

Based on its review as set forth above, the NRC staff finds that Salem will continue to meet the requirements of GDC 15, "Reactor Coolant System Design." The RCS and associated auxiliary, control, and protection systems are designed with sufficient margin to ensure that the design conditions of the RCS boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences, at the uprated power level of 3,459 MWt. Therefore, the staff finds this acceptable.

3.1.3 Steam Generators

3.1.3.1 Model F Steam Generator Structural Integrity (Unit No. 1)

In 1998, PSEG replaced the original Model 51 steam generators in Salem Unit No. 1 with Westinghouse Model F steam generators. PSEG determined that the design parameters for the Model F steam generators were the same or more conservative than the parameters specified for the original Model 51 steam generators. Consequently, the previous analysis of design transients and resulting fatigue cumulative usage factors remain valid for the Model F steam generators. The 1.4% power uprate results in slight changes to the steam generator operating parameters; however, they are still bounded by the current design parameters. PSEG concluded that the current design basis structural and fatigue analyses for the Unit No. 1 steam generators remain valid for the power uprate. Therefore, since the current analyses remain bounding, the NRC staff concurs with PSEG's conclusion.

3.1.3.2 Model 51 Steam Generator Structural Integrity (Unit No. 2)

Westinghouse Model 51 steam generators are currently installed in Salem Unit No. 2. PSEG evaluated the power uprated design conditions such as primary and secondary pressures and temperatures that affect the structural performance of the steam generator components. PSEG compared the uprated parameters to the parameters for the existing steam generator structural

and fatigue analysis. PSEG determined that the existing design basis analysis bounds the power uprated conditions, and the analysis of design transients is not affected by the power uprate. PSEG concluded that the current structural and fatigue analyses for the Unit No. 2 steam generators remain valid. Similarly, since the current analyses for the Model 51 steam generators remain bounding, the NRC staff concurs with PSEG's conclusion.

3.1.3.3 Steam Generator U-Bend Wear

U-bend wear can lead to the plugging of steam generator tubes. The highest power level and the lowest steam pressure characterize the limiting condition for U-bend wear.

The current design basis operating conditions of the Model 51 steam generators in Salem Unit No. 2 bound the operating conditions at the proposed 1.4% increase in power level. For the Model F steam generators of Salem Unit No. 1, the current design parameters of the Unit No. 1 steam generators bound the expected operating parameters at the proposed 1.4% higher power level. Therefore, the proposed increase in power level by 1.4% will have little or no effect on the expected U-bend wear rates of the steam generators.

PSEG also determined that the 1.4% increase in power and minimum steam pressure conditions are bounded by the conditions analyzed in the current design basis analysis. The results of the design basis analysis showed that only one additional tube per steam generator would be subject to plugging as a result of long-term operation at those conditions. Since the limiting conditions for the power uprate are considerably less severe than the design basis conditions, the staff concurs with the licensee's conclusion that increased U-bend wear is not significant, and finds this acceptable.

3.1.3.4 U-Bend Fatigue Evaluation

PSEG performed an analysis to assess the impact of the 1.4% power increase on steam generator U-bend fatigue. The Salem Unit No. 2 steam generator tubes are the most susceptible to U-bend fatigue, since the requisite conditions for the development of high cycle fatigue cannot occur in the replacement Model F steam generators currently in Salem Unit No. 1. Based on PSEG's analysis, the tubes that would require preventive action for Salem Unit No. 2, at the given steam pressures, have been identified. PSEG performed a screening process of the critical parameters associated with the susceptibility of the limiting steam generator tubes. The bounding values were identified and found adequate. For those tubes which may become more susceptible to a tube rupture for steam pressures below 650 psia, administrative controls will be put in place to ensure that the effects of secondary side pressure are properly addressed. The staff notes that, at a steam pressure of 650 psia, the main turbine throttle valves would have reached the valves-wide-open point, and there are only 2 tubes that would be susceptible at this pressure that have not been plugged as a preventive measure. In view of the conditions necessary for steam pressure to decline below 650 psia, and the number of susceptible tubes, the administrative controls described in PSEG's November 10, 2000, application appropriately address concerns about the effects of secondary side pressure. Therefore, the staff concurs that the proposed increase in power level by 1.4% has little or no effect on the expected U-bend fatigue of the steam generator tubes.

3.1.3.5 Evaluation of Steam Generator Tube Degradation Mechanisms

Primary Water Stress Corrosion Cracking (PWSCC)

In its assessment of the replacement steam generators in Unit No. 1, PSEG stated that the thermally treated Alloy 600 tubing in the replacement steam generators has shown higher resistance to stress corrosion cracking than mill-annealed Alloy 600 tubing in the original steam generators. PSEG determined that the power uprate will have a negligible impact on stress corrosion cracking initiation.

In its March 28, 2001, letter, PSEG identified the following active degradation mechanisms following the Unit No. 2 steam generator tube examinations during refueling outage 11: (1) PWSCC at the hot leg tubesheet expansion transition zone; (2) PWSCC at hot leg dented tube support plate intersections; (3) PWSCC in low row U-bends; (4) PWSCC in Alloy 600 tube plugs; (5) ODSCC in the hot leg sludge pile region; (6) anti-vibration bar (AVB) wear; and (7) cold leg thinning.

It is known that the T_{hot} will affect the aforementioned stress corrosion cracking of the steam generator tubes. The power uprate will result in an increase in the design T_{hot} of $0.5\text{ }^{\circ}\text{F}$. PSEG used the Arrhenius Equation with a conservative T_{hot} increase of $1\text{ }^{\circ}\text{F}$ to assess the sensitivity of the T_{hot} increase on steam generator tube stress corrosion cracking. From the assessment, PSEG predicted that ODSCC growth rates will increase by 2 to 3%, and PWSCC growth rates will increase by 3 to 4%. PSEG has incorporated these growth rate changes in the Unit No. 2 Operating Cycle 12 operational assessment. As a result, the staff concludes that incorporating this information into PSEG's future tubing examination program is appropriate.

In its sensitivity assessment, PSEG also considered steam pressure fluctuations that have a secondary effect upon stress corrosion cracking growth rates. As a result of the power uprate, the secondary side pressure will also be reduced by 5 to 6 psi. PSEG further determined that this pressure reduction will have an insignificant effect on operating tube stresses. The staff concurs with the licensee's evaluation and finds this acceptable.

Adverse secondary-side chemistry conditions will affect ODSCC and cold leg thinning in steam generator tubing. PSEG has improved the secondary side chemistry, which is expected to balance any potential impact upon ODSCC growth rates from the power uprate. PSEG concluded that cold leg thinning is not expected to be affected by the T_{hot} increase because cold leg thinning has been associated with localized crevice chemistry conditions. These conditions have been addressed by chemical cleaning during Salem Unit No. 2 refueling outage 10 and by improvements implemented in the secondary side chemistry control program. The staff finds that the proposed power increase will have minimal impact on ODSCC concerns and that PSEG is taking appropriate action to address this issue.

Anti-Vibration Bar (AVB) Wear

PSEG stated that steam pressure reduction and flow rate increases in the secondary side can affect the AVB wear growth rates, if the changes in these parameters are significant. PSEG determined that the changes due to power uprate are small; therefore, the power uprate has a

negligible impact on the AVB wear growth rates. In addition, AVB wear is monitored during each outage and site-specific AVB wear growth rates are calculated and are used to develop the operational assessment.

In the Unit No. 1 replacement steam generators, PSEG has observed AVB wear, which is consistent with other domestic Model F steam generators. As part of its condition monitoring and operational assessments, PSEG has evaluated AVB wear growth rates following each outage. The Unit No. 1 cycle 15 operational assessment will address the effects of uprate on applicable performance criteria.

PSEG concluded that the power uprate conditions will have a negligible impact on the existing degradation mechanisms. On the basis of the information presented, the NRC staff agrees with this conclusion.

Tube Plugging Limits and Inspection Frequencies

In addition to tube degradation mechanisms, the NRC staff also reviewed the potential impact of the power uprate on tube plugging limits and tube inspection frequency. These topics are discussed below.

The NRC staff questioned whether under the power uprate conditions the current 40% through-wall plugging limit for the steam generator tubes in the Salem Unit Nos. 1 and 2 TSs conforms with NRC Regulatory Guide (RG) 1.121. RG 1.121 recommends that steam generator tubing maintain certain safety margins under normal operating and faulted conditions to demonstrate its structural integrity. PSEG determined that plant parameters that affect degradation growth rates, such as T_{hot} , will change minimally under power uprate conditions.

PSEG stated that under certain situations, when flaw growth rates would not support operational assessment performance criteria for the planned operational period following an inspection outage, tubes will be plugged at less than the current 40% TS limit. If the plugging limit were not lowered below the current 40% plugging limit for a specific degradation mechanism and operational assessment performance criteria could not be met for the proposed operating period, PSEG would conduct SG tube inspections more frequently. Also, if PSEG chooses to extend the cycle run time to support the normal 18-month refueling frequency, in some cases an administrative plugging limit below 40% may be implemented. PSEG determined that the power uprate will have an insignificant effect on existing and anticipated degradation growth rates, and safety limits will continue to be maintained. PSEG concluded that, under the power uprate conditions, the 40% through wall tube plugging limit in the Salem Unit Nos. 1 and 2 TSs satisfies RG 1.121. The NRC staff concurs with this conclusion.

The NRC staff questioned whether the power uprate will affect the scope and frequency of steam generator tube inspections. PSEG stated that the steam generator tube inspection scope and frequency are driven by the degradation assessment, condition monitoring and operational assessments, Electric Power Research Institute (EPRI) steam generator inspection guidelines, and TS requirements. Tube inspections are controlled by evaluating both active and potential degradation mechanisms, industry experience, and plant-specific operating experience. As previously noted, the power uprate will have minor effects on observed growth rates of existing degradation mechanisms and is not expected to create new degradation mechanisms. PSEG concluded that no changes to the inspection scope and frequencies are

required under the power uprate conditions. Given that the changes in steam generator operating parameters T_{hot} and T_{cold} are small, the NRC staff finds this conclusion acceptable.

3.1.3.6 Steam Generator Evaluation Conclusion

Based on the information provided and its review, the staff concludes that the steam generators will continue to meet the requirements of GDC 14 and 15 to ensure that the reactor coolant pressure boundary and associated auxiliary systems have been designed, fabricated, erected, and tested so as to have a low probability of abnormal leakage, rapid failure and gross rupture during normal operation and anticipated operational occurrences.

3.1.4 Chemical and Volume and Control System (CVCS)

The primary role of the Chemical and Volume Control System (CVCS) is to manage reactor coolant system (RCS) water inventory, boron concentration, and water chemistry. In order to meet these requirements, the CVCS has to perform the following functions: (1) add boric acid and corrosion control chemicals to the reactor coolant; (2) cleanup, degasify, and makeup reactor coolant; and (3) reprocess the letdown water from the RCS and the water from reactor coolant pump (RCP) seal water injection. The power uprate will change the operating temperatures of the RCS, which could have some impact on these CVCS functions. However, PSEG's analysis has indicated that after power uprate the maximum cold leg temperature will be 542.7 °F which is lower than the design system inlet temperature (i.e., conservative), and much lower than the shell side design temperature for the regenerative heat exchanger. Also, since the excess letdown path is used to process excess coolant caused only by its expansion during plant heatup, it will not be affected by the revised cold leg temperature at full power operation. At full power operation the desired outlet temperature and flow for the excess path could be maintained by throttling the excess letdown heat exchanger outlet flow. PSEG concluded, therefore, that the proposed power uprate will not cause any deleterious effects to the operation of the CVCS. The staff has reviewed PSEG's analysis and, for the reasons presented by the licensee, concludes that the CVCS operation will be unaffected by the power uprate.

3.1.5 Safety Injection System (SIS)

The concludes that the revised design conditions as a result of the power uprate will have no direct effect on the overall performance capability of the SIS. In addition, the high and low pressure SIS systems will continue to deliver flow at the design basis RCS and containment pressures since there are no changes in the RCS operating pressure. Therefore, the staff finds this acceptable.

3.1.6 Residual Heat Removal System (RHR)

PSEG reviewed single train cooldown and normal cooldown cases to address the uprated power conditions. PSEG concluded that both the normal and single train cooldown can be accomplished within 36 hours at the power uprate conditions, which meets the RHR design condition. The staff concurs with PSEG's analysis and finds this acceptable.

3.1.7 Component Cooling Water System (CCWS)

The CCWS serves as an intermediate system between the ultimate heat sink and the different plant systems carrying radioactive fluids. After the power uprate, decay heat transferred from the RHR system and from the spent fuel pool to the CCWS will increase. However, PSEG's accident analysis for LOCA, which is a postulated limiting accident, was performed for 102% of the reactor power. Therefore, this analysis bounds the proposed power uprate. The power uprate will also cause a small increase in decay heat. Although it will produce a slightly longer single train cool down time to 200 °F, system cool down still could be accomplished within the required time interval. The staff has reviewed PSEG's analysis and finds that, since the changes in heat loads coming from components served by the CCWS are small, the operation of the CCWS will not be affected by the power uprate. Therefore, CCW will continue to be capable of performing its' safety function.

3.1.8 Waste Disposal System (WDS)

The WDS consists of separate gaseous and liquid waste processing systems. PSEG's evaluation indicates that these systems will not be affected by the power uprate. The staff concurs with PSEG's finding.

3.1.9 Sampling System (SS)

The role of the SS is to provide various flow paths for taking samples from the RCS and selected auxiliary systems. These samples are drawn and cooled before being analyzed. The maximum hot leg temperature after power uprate will not exceed 613.1 °F which is much lower than the sampling system's heat exchanger design temperature of 653 °F. PSEG concluded that after power uprate the design performance of the system will not change. Since the hot leg temperature will continue to be less than the design temperature, the SS will continue to be adequate for operation under uprated conditions. Accordingly, the staff finds this acceptable and concurs with PSEG's conclusion.

3.1.10 Containment Spray System (CSS)

The uprated conditions have no direct impact on CSS performance capability. Thus, the staff concludes that the proposed power uprate is acceptable with respect to the CSS.

3.1.11 Nuclear Steam Supply System (NSSS) Auxiliary Equipment

Auxiliary equipment includes heat exchangers, pumps, valves, and tanks in the auxiliary systems. Fatigue analysis of this equipment showed that fatigue usage will remain bounded by the current analysis and less than 1.0. Therefore, the staff find that the amendments are acceptable with respect to this equipment.

3.2 NSSS/Balance-of-Plant Interface Systems Evaluation

3.2.1 Main Steam System (MSS)

The major components of the MSS include the steam generator main steam safety valves (MSSVs), the steam generator PORVs and the main steam isolation valves (MSIVs).

PSEG evaluated the effects resulting from plant operations at 101.4% of the current rated thermal power on the MSS, including the steam generator MSSVs, the steam generator PORVs, the MSIVs, and the MSIV bypass valves. Based on the discussions in the following sections, PSEG concluded that the components are adequately sized for operation at proposed uprated power levels. PSEG determined that no hardware or operational modifications are required and that the current design basis remains valid.

3.2.1.1 Steam Generator Main Steam Safety Valves

According to the Salem FSAR, the MSS piping is in compliance with ANSI B31.1, and was designed to the appropriate wall thickness formula from the 1965 Edition and Summer of 1966 Addenda of the ASME Code. The MSS has 20 safety valves with a total capacity of 16.65×10^6 lb/hr in each operating unit. This provides about 110.3% of the maximum calculated steam flow of 15.10×10^6 lb/hr for the revised design conditions for Unit No. 1, and 110.4% of the maximum calculated steam flow of 15.08×10^6 lb/hr for Salem Unit No. 2. Thus, based on the range of NSSS performance parameters for the uprating, the capacity of the installed MSSVs meets the appropriate Code sizing requirements, and the staff finds this acceptable.

3.2.1.2 Steam Generator Power-Operated Relief Valves

The steam generator PORVs are sized to have a capacity equal to 10% of the steam flow used for plant design, at no-load steam pressure. For the revised design conditions, each steam generator PORV is required to have a capacity equal to 382,794 lb/hr at 1020 psia steam pressure. At these conditions, this capacity permits a plant cooldown to the RHR system operating conditions in 4 hours assuming a minimum of 2 hours at hot standby. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the auxiliary feedwater system. Since the design capacity of the installed PORVs meets the analyzed sizing criteria, the PORVs are adequately sized for the 1.4% uprated conditions.

3.2.1.3 Main Steam Isolation Valves and Main Steam Isolation Bypass Valves

Rapid closure of the MSIVs following postulated steamline breaks causes a significant differential pressure across the valves' seats and a thrust load on the main steam system piping and piping supports in the area of the MSIVs. The worst cases for differential pressure increase and thrust loads are controlled by the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since these variables and no-load operating pressure are not impacted by the uprating, the design loads and associated stresses resulting from rapid closure of the MSIVs bound potential challenges due to the power uprate. Consequently, the proposed power uprate has no significant impact on the interface requirements for the MSIVs.

The MSIV bypass valves are used to warm up the main steamlines and equalize pressure across the MSIVs prior to opening the MSIVs. The MSIV bypass valves perform their function at no-load and low-power conditions where the proposed power uprate has no significant impact on main steam conditions (steam flow/steam pressure). Therefore, the proposed power uprate has no significant impact on the interface requirements for the MSIV bypass valves.

3.2.1.4 Steam Dump System

The steam dump system provides a steam dump capability of about 43.8% of the original maximum guaranteed steam flow at a full-load steam generator pressure of 805 psia versus the sizing criterion of 40% of rated steam flow. Operation of the NSSS within the proposed range of operating parameters at increased steam flow will result in a small decrease in steam dump capacity. Based on the range of NSSS operating parameters approved for power uprate, the steam dump capacity continues to meet the design criterion. The condenser steam dump capacity is adequate for the proposed 1.4% power uprate.

PSEG evaluated the steam dump system for plant operations at 101.4% of the current rated thermal power and concluded that at the higher power level, the steam dump system capacity still exceeded the Westinghouse criterion to discharge 40% of the rated steam flow at full load steam pressure. PSEG determined that no hardware or operational modifications are required and that the current design basis remains valid. Therefore, the staff has determined that the 1.4% power uprate will have no impact on the operation of the steam dump system.

3.2.1.5 Main Steam System Evaluation Conclusion

The staff has reviewed PSEG's submittal, and, for the reasons stated above, concludes that the proposed 1.4% power uprate will have no adverse impact on the operation of the MSS. Therefore, the staff finds this acceptable.

3.2.2 Condensate and Feedwater Systems

PSEG evaluated the effects of plant operations at 101.4% of the current rated thermal power on the condensate and feedwater systems. The condensate and feedwater systems automatically maintain steam generator water levels during steady-state and transient operations. The revised design conditions will have a slight impact on the feedwater volumetric flow and system pressure drop. The major components of the systems are the main feedwater isolation valves (MFIVs), the feedwater regulator valves (MFRVs), and the condensate and feedwater pumps. PSEG determined that no hardware or operational modifications are required for the slight changes in feedwater volumetric flow and system pressure drop, and that the current design basis remains valid. Therefore, the staff has determined that the 1.4% power uprate will have no impact on the operation of the condensate and feedwater systems.

The MFRVs are designed to withstand the dynamic loads created from maximum feedwater flow following a steamline break from no load conditions. These loads far exceed any normal operational loading difference as a result of the power uprate. Therefore, these valves are not affected by the proposed power uprate. The condensate feedwater pumps are sized to provide 96% flow at full power with a 100 psi pressure above full load pressure. From the current design, the staff concludes that the pumps are sized to accommodate the 1.4% full power level increase.

PSEG's analysis of record used a power level of 102% to estimate the condensate tank capacity of 200,000 gallons. Therefore, the staff concludes that the condensate tank has been sized adequately for the 1.4% power uprate.

3.2.3 Auxiliary Feedwater System

As an engineered safeguards system, auxiliary feedwater (AFW) supplies feedwater to the secondary side of the steam generators when the normal feedwater system is not available, thereby maintaining the steam generator heat sink. The AFW pumps take suction from the auxiliary feedwater storage tank (AFST) to fulfill the engineered safety function during a transient to enable the plant to be placed in a safe shutdown condition. PSEG evaluated the effects of plant operations at 101.4% of the current rated thermal power on the AFW system. The original analyses for the limiting accident for the AFW system (double-ended feedwater line break) assumed a power level of 102% reactor power. Operation at 101.4% above current power is bounded by the original analysis.

As previously stated, the AFW pumps are normally aligned to the AFST. The AFST design function is to supply sufficient inventory during accident or transient conditions to enable the plant to be placed in a safe shutdown condition. The minimum inventory requirement is based on a reactor trip from 102% of rated thermal power. Thus, no change in the required inventory is required for operation at the uprated power level of 101.4%.

Based on its review, as set forth above, the staff concurs with PSEG's evaluation that operations at the proposed 1.4% uprated power level will have an insignificant impact on the operation of the AFW system. The staff's conclusion is based on its review of PSEG submittal, and the experience gained from the review of power uprate applications for similar pressurized water reactor plants.

3.2.4 Steam Generator Blowdown System

The function of the steam generator blowdown system is to control the chemical composition of the steam generator shell side water within specified limits and buildup of solids in the steam generator water. The blowdown flow rates during power operation are determined by the water chemistry and by the tube-sheet sweep required for controlling buildup of solids. The rate at which dissolved solids are introduced into the steam generator secondary water depends on condenser leakage, quality of the secondary makeup water, and the rate of generation of corrosion products in the secondary water system.

The proposed power uprate does not alter the first two sources of dissolved solids, and only the increase of generation of corrosion products may change the ingress of particulates into the steam generator water. Flow accelerated corrosion is the most important mechanism for generation of corrosion products. The rate at which corrosion products are generated by this mechanism depends on the velocity of flow of water in the affected systems. PSEG determined that the change of the flow after power uprate will be very small. Therefore, the staff has determined that the change in generation of corrosion products will be insignificant, and the performance of the blowdown system will not be impacted.

The flow control valves in the blowdown system are designed for a specified range of inlet pressures between no load and full load plant operation. PSEG determined that these pressures will not change and there will be no need to modify these valves. PSEG concluded that the proposed power uprate will not produce any effect which would downgrade the operation of the steam generator blowdown system. The staff has reviewed PSEG's analyses

and concurs that the steam generator blowdown system will continue to perform its function of controlling the buildup of solids on the shell side of the steam generators following the proposed power uprate.

3.2.5 Containment Integrity Analyses

PSEG evaluated the short- and long-term LOCA and Steamline Break mass and energy releases with respect to the 1.4% power uprate and determined that a calorimetric uncertainty of 2% was incorporated into the analysis. The NRC approved CENP Crossflow UFM system topical report, CENPD-397-P-A, Revision 1, for referencing in license applications for power uprate in a safety evaluation dated March 20, 2000. The topical report was provided as the basis for reduction of the calorimetric uncertainty from 2% to 0.6%. CENP calculations for Salem indicate a flow accuracy of better than 0.5% of rated flow for the Salem site-specific installations of the Crossflow system. Based on this evaluation, there is no margin reduction associated with operation at 101.4% of the present rated core thermal power with a 0.6 percent calorimetric uncertainty when using the approved Crossflow instrumentation.

The change in the uncertainty allowance that is applied to the core power can affect only the initial power used in the analysis; all other conservative assumptions remain unchanged. The Analyses of Record applicable to both Salem units for the inside and outside containment long-term steamline breaks assume a 2% power calorimetric uncertainty on a 3,431 MWt NSSS power. A minimum 0.6% calorimetric uncertainty applied to a maximum 1.4% power increase is equivalent to the licensing basis Analyses of Record. The critical parameters for short-term steamline breaks are defined at no-load conditions, when the steam generator level and enthalpy are both high. As previously indicated, through the use of the improved Crossflow instrumentation, the requested increase in operating power level by 1.4% is based on reducing feedwater flow measurement uncertainty (a direct input to the calorimetric calculation) by the same 1.4%. In all cases, PSEG determined that the mass and energy release calculations remain valid, and that the containment integrity analyses are unaffected by the proposed 1.4% uprate.

Based on its review, as set forth above, the staff concurs with PSEG's evaluation that operations at the proposed 1.4% uprated power level will have an insignificant impact on containment integrity. Therefore, the staff concludes that the requirements of GDC 16, "Containment Design," and GDC 50, "Containment Design Basis," will continue to be met.

3.3 Balance-of-Plant Systems

PSEG evaluated the following Balance-of-Plant (BOP) systems for operation at the 1.4% power uprate:

- Feedwater
- Main Turbine
- Condenser
- Condensate
- Heater Drain
- Service Water
- Circulating Water
- Turbine Auxiliary Cooling

- Heating, Ventilation, and Air Conditioning
- Component Cooling Water
- RHR Shutdown Cooling
- Spent Fuel Pool Cooling
- (Radioactive) Waste Disposal
- Motor-Operated Valves

3.3.1 Balance of Plant Systems Evaluation

3.3.1.1 General

PSEG evaluated the adequacy of the BOP systems based on comparing the existing design bases parameters with the core power uprate conditions.

PSEG further evaluated the affected systems on the basis of the uprated input parameters shown in Tables 2-1 and 2-2 of Attachment 1 to the November 10, 2000, application for RCS temperatures, steam temperature and flow rate, and the heat balance at 3,459 MWt reactor power. As a result, PSEG concluded that the existing design basis analyses, using maximum differential temperatures and pressures for normal operation and worst case conditions, for the BOP piping, pipe supports, and components remain bounding for the uprated power level of 3,459 MWt at Salem.

The BOP systems were designed for the turbine valves wide-open condition, corresponding to an NSSS power of approximately 3,570 MWt, a power in excess of the proposed uprate. No BOP hardware changes, and no significant setpoint changes were identified, because the uprate should be accommodated within the excess capacity of the as-designed BOP equipment. The only hardware changes are those required for adding the proposed Crossflow flow measurement system. This system does not replace the present feedwater flow instrumentation that provides continuous control room indication and feedwater flow control. The new Crossflow measurement system is used to periodically perform the calorimetric calculation and calibrate the power range meters.

PSEG also addressed the thermal-hydraulic performance, piping and support qualification, instrumentation and control functionality, equipment performance, and impact on the existing radiological consequences of a pipe break. PSEG determined that no hardware or operational modifications are required, and in all cases the current design basis remains valid. Only design documentation changes and minor instrumentation and control set point-related changes are necessary.

In addition, PSEG evaluated high energy line breaks outside of containment. The evaluation concluded that the 1.4% power increase will have no effect on either the current licensing basis steamline break mass and energy release analysis, or the FSAR conclusions with regard to line breaks.

Based on its review, for the reasons set forth above, the NRC staff finds that the BOP will continue to meet its design requirements and is acceptable for the proposed power uprate.

3.3.1.2 Motor-Operated Valves (MOVs)

In its November 10, 2000, application, PSEG stated that there are no changes to its MOV program as a result of the proposed 1.4% power uprate. The safety-related valves were not impacted by the proposed increase in licensed power levels, and are, therefore, acceptable. This determination was confirmed by verifying that changes in system operating temperature, pressure, and flow rate were bounded by the requirements of the associated equipment specifications. As such, the increased thrust required to operate the MOVs due to expected differential pressure conditions is within the capabilities of the existing valve actuators. Additionally, in its letter dated April 20, 2001, PSEG indicated that its MOV design calculations are based on the limiting condition (highest pressure differential), which occurs at the no-load condition. Consequently, the proposed power uprate does not impact the GL 89-10 MOV program.

PSEG reviewed the evaluation of GL 95-07 associated with the pressure locking and thermal binding of safety-related power operated gate valves. PSEG found that the existing analysis conditions remain bounding for the proposed 1.4% power uprate. PSEG further reviewed the evaluation of its GL 96-06 program regarding the over-pressurization of isolated piping segments. PSEG concluded that the existing evaluation for GL 96-06 was performed at 102% of 3,411 MWt, and is bounding for the proposed power uprate of 101.4% of current rated power. On the basis of its review, for the above reasons, the NRC staff concurs with PSEG's conclusions that the power uprate will have no adverse effects on the safety-related valves and that the conclusions of PSEG's GL 95-07, GL 96-06, and GL 89-10 programs remain valid.

3.3.2 NRC Staff's Conclusion

The NRC staff concurs with PSEG that operations at the proposed 1.4% uprated power level of 3,459 MWt will have an insignificant impact on the balance of plant systems. Therefore, based on its review, as set forth above, the staff concludes that the BOP piping, pipe supports and equipment nozzles, and valves remain acceptable and continue to satisfy the design-basis requirements for the power uprate.

3.4 Instrumentation and Control Systems

3.4.1 CENP Crossflow UFM

3.4.1.1 Background

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the NSSS energy balance. This calculation is called "secondary calorimetric" for a pressurized water reactor (PWR) and "heat balance" for a boiling water reactor (BWR). The accuracy of these calculations depends primarily upon the accuracy of feedwater flow and main steam and feedwater temperature and pressure measurements. Feedwater flow is the most significant contributor to the core thermal power uncertainty. An accurate measurement of these three parameters will result in an accurate determination of core thermal power and an accurate calibration of the nuclear instrumentation.

The instrumentation used for measuring feedwater flow is typically an orifice plate, a venturi meter, or a flow nozzle. These devices generate a differential pressure proportional to the

feedwater velocity in the pipe. Of the three differential pressure devices, a venturi meter is most widely used for feedwater measurement in nuclear power plants. The major advantage of a venturi meter is a relatively low head loss as the fluid passes through the device. The major disadvantage of the device is that the calibration of the flow element shifts when the flow element is fouled, which causes the meter to indicate a higher differential pressure and hence a higher than actual flow rate. This leads the plant operator to calibrate nuclear instrumentation high. Calibrating the nuclear instrumentation high is conservative with respect to the reactor safety, but causes the electrical output to be proportionally low when the plant is operated at its thermal power rating. To eliminate the fouling effects, the flow device has to be removed, cleaned, and recalibrated. Due to the high cost of recalibration and the need to improve flow instrumentation uncertainty, the industry assessed other flow measurement techniques and found the Crossflow UFM to be a viable alternative. The measurement uncertainties due to venturi fouling and instrumentation drift and calibration shifts are essentially eliminated when a Crossflow UFM is used. The Crossflow UFM does not replace the currently installed plant venturi, but provides PSEG an in-plant capability for periodically recalibrating the feedwater venturi to adjust for the effect of fouling. A unique advantage of the Crossflow UFM system is that it is installed external to the pipe in which flow is to be measured, thereby eliminating any possibility of compromising pressure boundary integrity.

The Crossflow UFM consists of four ultrasonic transducers mounted on a metal support frame which is clamped on the feedwater piping. There is one upstream and one downstream transducer station. Each station consists of one transmitting and one receiving transducer. The operation of a cross-correlation UFM is based on the fact that an ultrasonic beam traveling across fluid flowing in a pipe is affected (modulated) by the turbulence (eddies) present in the flowing liquid. When this modulated signal is processed, a random signal which is a signature of the flowing eddies can be obtained. The Crossflow UFM calculates the time a unique pattern of eddies take to pass between two sets of ultrasonic transducers and divides the known distance between the two sets of transducers by the calculated time to obtain the flow velocity. This measured velocity is not an average velocity (highest velocity is at the center of the pipe) and should be multiplied by the "Velocity Profile Correction Factor" (VPCF) to obtain the average velocity of the fluid flowing in the pipe.

3.4.1.2 Crossflow System at Salem

PSEG will install a Crossflow UFM system for feedwater flow measurement in both Salem units. The Crossflow UFM system consists of a Mounting/Transducer Support Frame with ultrasonic transducers, a signal conditioning unit (SCU), and a data processing computer (DPC). The DPC receives a feedwater flow signal from the SCU and feedwater pressure and temperature input from the plant computer. Using a built-in signal processing algorithm, the Crossflow DPC calculates fluid velocity in the common header and converts the fluid velocity to a mass flow using flow, temperature, and pressure as calculation inputs. The Crossflow feedwater mass flow is periodically compared to the feedwater venturi mass flow to determine an adjustment of the venturi flow coefficient for obtaining the corrected mass flow signal. This corrected mass flow is used in calculating core thermal power and thereby calibrating nuclear instrumentation in accordance with the plant Technical Specification requirements.

PSEG stated that, although the Crossflow system for this application is non-safety-related, the system is designed and manufactured under the vendor's quality control program. This program provides for configuration control and maintenance and deficiency reporting and

correction, and the current software was verified and validated under CENP's verification and validation program. PSEG also stated that: (1) the Crossflow system will be included in the plant preventive maintenance program; (2) technical support personnel will monitor its reliability; (3) the equipment problems will be documented and corrected in accordance with PSEG's corrective action program; and (4) the system software is subject to PSEG's software quality assurance program.

CENP Topical Report CENPD-397-P-A (previously approved by the staff) describes the Crossflow UFM system for the measurement of feedwater flow and provides a basis for the proposed 1.4% uprate of the licensed reactor power. The topical report stated that the Crossflow UFM system is able to achieve an uncertainty of 0.5% or better with 95% confidence. The topical report provides specific guidelines and equations for determining uncertainty values of the Crossflow input parameters (VPCF, inside diameter, transducer spacing, feedwater density, and Crossflow time delay). The plant-specific uncertainties are determined when the meter is installed, using the guidelines and equations provided in the topical report. The topical report stated that a trained CENP representative will install the hardware and software of the Crossflow UFM. The topical report calculation showed Crossflow measurement uncertainty in a typical feedwater loop (straight pipe, fully developed flow) to be less than 0.5%. PSEG's submittal referenced CENP calculations A-SA1-PS-0001 and A-SA2-PS-0001, "Feedwater Flow Measurement Using the Crossflow UFM at PSEG Salem Units 1 and 2", for the plant-specific feedwater measurement uncertainty. The staff performed an audit review of A-SA2-PS-0001 and found the calculation established a 0.47% measurement uncertainty of the Crossflow UFM with 95% confidence for the Salem site-specific installation.

Since the Crossflow measurement uncertainty is affected by temperature change, the topical report recommended improving the accuracy of the feedwater temperature instrumentation. In the Salem Units 1 and 2 design, feedwater temperature is measured by RTDs through the computer readout. PSEG's submittal stated that to support the installation of Crossflow UFM, four additional computer inputs of feedwater temperature were added. These computer inputs will be checked on a monthly basis and recalibrated, if found to be out of tolerance. An annual calibration check will also be performed and new computer coefficients for the feedwater RTDs will be established, if plant computer indications were found out of tolerance. The RTDs are calibrated by heating them with a high accuracy dry heat block and combining the result with the instrument drift specified by the RTD vendor. PSEG's calculation found the feedwater instrumentation measurement uncertainty to be within 20 F° of the 434.6 °F nominal setpoint value. The staff believes that PSEG's monthly check and calibration program will adequately maintain the calculated uncertainty of the feedwater temperature input to the Crossflow UFM.

3.4.1.3 CENP Topical Report CENPD-397-P-A Criteria

The NRC staff's SER on CENP Topical Report CENPD-397-P-A included four additional criteria to be addressed by a licensee requesting power uprate. PSEG's submittals addressed each of the four criteria as follows:

- (1) "The licensee should discuss the development of maintenance and calibration procedures that will be implemented with the Crossflow UFM installation. These procedures should include processes and contingencies for an inoperable UFM and the effect on thermal power measurement and plant operation."

In its letter dated April 20, 2001, PSEG stated that calibration and maintenance of the Crossflow UFM will be performed using site procedures developed from the Crossflow system technical manuals. The current software was verified and validated under CENP's Verification and Validation Program, and a periodic online monitoring of the Crossflow system will verify that the SCU, DPC, and software remain within the stated accuracy. The Crossflow and plant computer software, and hardware configuration will be controlled by plant procedures applicable to the digital plant instrumentation. Plant personnel will be trained by the Crossflow system vendor to perform diagnostics, maintenance, and calibration of Crossflow UFM.

PSEG also stated that the Crossflow system does not perform any safety functions and is not used to directly control any plant systems. Therefore, should the Crossflow system become inoperable, it would have no immediate effect on thermal power measurement uncertainty or plant operation. Crossflow system inoperability is automatically transmitted to the plant computer to alarm an overhead annunciator in the control room. In this condition, Salem procedures will provide that the plant may continue to operate at the uprate power level for 24 hours and then reduce the operating power to the previously licensed power level if the Crossflow system does not return to an operable status. The plant will continue to operate at the reduced power level until the Crossflow system is restored to service, and a heat balance is performed in accordance with the plant TS requirement using the venturi measurement and an updated correction factor from the Crossflow system. The equipment problems will be documented and corrected in accordance with PSEG's Corrective Action Program. In addition, the system software is subject to PSEG's software Quality Assurance program.

In response to the staff's request for additional information (RAI), PSEG described its programs for the calibration of all other instrumentation in addition to the Crossflow, whose measurement uncertainties affect the plant power calorimetric uncertainty. These other instruments are for feedwater pressure and temperature, steam pressure, and steam generator blowdown flow. PSEG identified plant procedures applicable to these instruments and specified their calibration intervals. PSEG also listed the applicable procedures for performing corrective actions, reporting deficiencies to the manufacturers, and receiving and addressing manufacturer deficiency reports on these instruments. The staff concludes that PSEG's plant procedures can sufficiently ensure instrumentation capability to provide acceptable power calorimetric uncertainty for the proposed power uprate.

- (2) "For plants that currently have Crossflow UFM installed, the licensee should provide an evaluation of the operational and maintenance history of the installed UFM and confirm that the instrumentation is representative of Crossflow UFM and bounds the requirements set forth in Topical Report CENPD-397-P-A."

Because Crossflow UFM is not currently installed at Salem, Unit Nos. 1 and 2 to achieve the 1.4% power uprate. No plant-specific maintenance and operational data are available for this evaluation.

- (3) "The licensee should confirm that the methodology used to calculate the uncertainty of the Crossflow UFM in comparison to the current feedwater flow instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument

uncertainty). If an alternate methodology is used, the application should be justified and applied to both the venturi and the UFM for comparison.”

In its November 10, 2000, application, PSEG stated that the Crossflow UFM measurement uncertainty calculations (calculations A-SA1-PS-0001 and A-SA2-PS-0001) are based on an accepted plant instrument uncertainty methodology and are consistent with the methodology described in Topical Report CENPD 397-P-A, Revision 01. PSEG’s submittal also included the Crossflow assisted power calorimetric uncertainty calculation Westinghouse Topical Report WCAP-15553, “Power Calorimetric for the 1.4% Upgrading for Public Service Electric and Gas Company Salem Units 1 and 2.” This calculation used a conservative value of 0.5% for the Crossflow as a component of the plant power calorimetric uncertainty. The staff’s review found that WCAP-15553 accounted for all components of the plant calorimetric measurement uncertainty and followed the approved setpoint methodology for calculating the power measurement uncertainties. The total power measurement uncertainty in the plant calorimetric calculation was found to be 0.6% to support the proposed 1.4% power uprate.

- (4) “The licensee of the plant at which the installed Crossflow UFM was not calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation), should submit additional justification. This justification should show that the meter installation is either independent of the plant-specific flow profile for the stated accuracy or that the installation can be shown to be equivalent to known calibrations and plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed and calibrated Crossflow UFM, the licensee should confirm that the piping configuration remains bounding for the original Crossflow UFM installation and calibration assumptions.”

PSEG stated in its November 10, 2001, letter that the Salem Crossflow system will be installed and calibrated to the site-specific piping configuration (with the flow profile and meter factors representative of the plant-specific installation) and the installation will follow the guidelines in the Crossflow UFM topical report. The staff’s audit review of Westinghouse Calculation A-SA2-PS-0001 showed that the calculation used plant-specific input parameters.

The staff finds that PSEG’s response to these criteria has sufficiently resolved the plant-specific concerns about Crossflow UFM maintenance and calibration, hydraulic configuration, and procedures and contingency plans for an inoperable Crossflow. PSEG used an approved methodology to calculate the plant-specific Crossflow measurement uncertainty and the plant power calorimetric measurement uncertainty.

3.4.1.4 Crossflow UFM Evaluation Conclusion

Based on a review of PSEG’s submittals, plant-specific Westinghouse calculations of Crossflow measurement uncertainty and plant power calorimetric measurement uncertainty, the staff finds that the Salem thermal power measurement uncertainty using the Crossflow UFM is limited to 0.6% of actual reactor thermal power and can support the proposed 1.4% thermal power uprate. The staff also finds that PSEG adequately addressed the four additional criteria outlined in the staff SER on the Crossflow Topical Report CENPD 397-P-A.

3.4.2 Engineered Safety Features Actuation System (ESFAS)

Normal operating transients are evaluated to confirm that the plant can appropriately respond to these transients without generating a reactor trip or engineered safety feature actuation system actuation. These are:

- 10 % step load increase
- 10 % step load decrease
- 50 % load rejection
- 5 % per minute ramp load increase

The analysis methodology for these transients employs a 2% power calorimetric uncertainty to increase the power level to 102%. The reduction in power measurement uncertainty and conservative assumptions provide substantial conservatism such that the transients noted above can be accommodated without resulting in a reactor trip or ESFAS actuation.

3.5 Electrical Systems

PSEG stated in its November 10, 2000, application that no alternating current (ac) or direct current (dc) auxiliary load ratings are expected to change, and that the loads are not expected to experience additional demands above their ratings. PSEG also stated that the main generator electrical parameters remain the same, and the uprate capacity remains within the generator rating. Voltage controls and grid source impedance at the Pennsylvania, New Jersey and Maryland Interconnection (PJM) 500kV grid will not be affected by this uprate, and the staff concludes that the evaluated voltages and short circuit values at different levels of the station auxiliary electrical distribution system will not change as result of the uprate.

GDC 17 of 10 CFR Part 50, Appendix A is the applicable criterion governing the requirements of the electric power systems. It requires, in part, that the onsite power system provide sufficient capacity and capability to support its specified safety functions. This is achieved by ensuring the onsite system has sufficient capacity and capability to supply the electrical loading required by the safety loads, and the voltage to those loads is adequate. The staff evaluated PSEG's statement that no ac or dc auxiliary load ratings are expected to change. The staff concludes that the capacity and capability of the onsite system to supply the required safety systems will not be impacted by this power uprate, and GDC 17 will continue to be met in this area.

GDC 17 also requires, in part, that the onsite power systems have sufficient independence and redundancy to perform their safety function assuming a single failure. Electrical protective devices with sufficient fault current interrupting capability are typically required to ensure a single fault will not cascade up to a level that would degrade redundant safety systems. PSEG's statements that the main generator electrical parameters remain the same and that the voltage controls and grid source impedance at the PJM 500kV grid will not be affected by the power uprate supports a conclusion that short circuit values will not change. The staff concludes that single failure requirements of GDC 17 will, therefore, continue to be met in this area.

GDC 17 further requires that provisions be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power

generated by the nuclear unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. In this regard there might be some potential that the stability of the transmission network could be affected by the increase in power generated by the Salem nuclear unit. In response to a staff question, PSEG stated in an April 20, 2001, letter that the review of the stability analysis by the PJM Interconnection LLC for the increase in power level has been completed. PSEG's review indicated that, for one of the cases evaluated, the operating parameters for PSEG will require minor changes to the minimum megavolt-amps, reactive (MVAR) limits due to the increase in power level. PSEG stated that the Artificial Island Operating Guide which controls the megawatt (MW) and MVAR operating curves specified by the PJM Interconnection for Salem and Hope Creek will be revised to incorporate these changes as part of the implementation plan for the increased power level. The staff finds these provisions acceptable to minimize the probability of losing power from the transmission network and is consistent with the requirements of GDC 17.

3.5.1 NRC Staff's Conclusion

Based on its review as set forth above, the staff finds that the proposed 1.4% power increase will not result in any deviations from the requirements of GDC 17 and is, therefore, acceptable from the perspective of the electric power systems.

3.6 Nuclear Steam Supply System Accident Evaluation

3.6.1 Steam Generator Tube Rupture Evaluation

The proposed power uprate increases the break flow across the steam generator by decreasing the secondary side pressure. In addition, by increasing the system energy, the power uprate also increases the amount of steam released from the ruptured steam generator. The higher break flow and steam releases could possibly result in higher offsite dose consequences. However, the current licensing basis analysis includes a 104.5% reactor power for calculating the steam release, which is then used for the offsite dose calculation. Thus, the current 104.5% reactor power calculation bounds the 1.4% power uprate.

3.6.2 Steamline Break Evaluation

3.6.2.1 Long-Term Steamline Break Mass and Energy Releases Inside and Outside Containment

Salem's Analyses of Record applicable to both units for the inside and outside containment long-term steamline breaks use 102% power as the initial power level. Thus, the power increase of 1.4% with a maximum uncertainty of 0.6% is bounded by the Analyses of Record.

3.6.2.2 Short-Term Steamline Break Mass and Energy Releases

The worst case short-term steamline break event occurs at no-load conditions. Therefore, a power uprate of 1.4% has no effect on the bounding analysis.

3.6.2.3 Radiological Steam Releases for Dose Calculations

The current radiological steam release calculations use 104.5% power, which bounds the proposed uprate for Unit Nos. 1 and 2. Steam releases from Salem Unit No. 1, with Model F steam generators that have a lower mass inventory than the Model 51 steam generators used in the analysis, are also bounded by the analysis.

3.6.3 LOCA Mass and Energy Releases

3.6.3.1 Long-Term LOCA/Containment Integrity Analysis

The most limiting Salem LOCA mass and energy release calculation was performed in accordance with the methodology in WCAP-8264-P-A. This methodology assumes a rated power of 3,570 MWt (approximately 4.3% greater than licensed power) and an additional 2% power uncertainty. This analysis bounds the 1.4% uprate. A separate analysis was performed for the replacement Model F steam generators of Unit No. 1. The analysis assumed a 3,411 MWt power with a 2% uncertainty, which also bounds the 1.4% uprate.

3.6.3.2 Short-Term LOCA Mass and Energy Release Analysis

The short-term LOCA mass and energy release calculations are performed to ensure that the walls in the immediate area of the break location can maintain their integrity during the short pressure pulse that accompanies a LOCA. The power uprate decreases the core inlet and vessel outlet temperatures, which in turn slightly increase the initial break flow rates into the reactor cavity and loop compartments. However, the reactor cavity and loop compartments are now analyzed under leak before break (WCAP-13659), rather than large break LOCA conditions. The reduction in break area associated with assuming a break in the largest branch line connected to the RCS primary loop, rather than a break in the main RCS piping, results in a decrease in mass and energy releases much greater than any change that can occur due to the power uprate effects. Therefore, the analyses, which are still based on breaks in the RCS main piping, remain bounding for the reactor loop compartment and reactor cavity region.

3.6.4 LOCA-Related Analyses

3.6.4.1 Large-Break LOCA and Small-Break LOCA

The current Large Break LOCA and Small Break LOCA accident analyses use an initial power level of 102%. The power increase of 1.4% with a corresponding 0.6% uncertainty is bounded by the current accident analyses.

3.6.4.2 Post-LOCA Long-Term Core Cooling (LTCC)

The current analysis indicates that the reactor will remain shutdown by using borated Emergency Core Cooling System Water from the RCS sump following a LOCA. The water volumes and the associated boric acid concentrations are not affected by the power uprate.

3.6.4.3 Hot Leg Switchover

The licensing basis hot leg switchover analysis uses a power rating of 102%, which includes a 2% uncertainty. A power increase of 1.4% with a maximum uncertainty of 0.6% remains bounded by the licensing basis analysis.

3.6.5 Reactor Vessel, Loop and Steam Generator LOCA Forces Evaluation

The proposed change of 1.4% with an uncertainty of 0.6% does not affect the magnitude of the heat sources mandated by 10 CFR Part 50, Appendix K for LOCA analyses. Thus, the existing LOCA hydraulic forces Analyses of Record supporting Salem Unit Nos. 1 and 2 will remain conservative.

3.6.6 Non-LOCA Transient Analyses

Trip Points and Time Delays

The non-LOCA transients were analyzed using statistical methods. The only quantities modified are the power level and the feedwater flow uncertainty. It was determined that the Over-Temperature Delta Temperature (OT Δ T) and the Over-Pressure Delta Temperature (OP Δ T) setpoints did not need to be modified to accommodate the power level increase with the new uncertainty. However, the Over-Temperature Δ T and f_1 (Δ I) function penalties will be revised slightly in the TSs for both Salem units.

3.6.6.1 Non-LOCA Transient Analyses Performed With Statistical Methods

Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power

The current Analysis of Record shows that approximately 14% safety analysis departure from nucleate boiling (DNB) margin exists, which is sufficient to offset an increase of 1.4% in peak power (1.4% power corresponds to an expected upper bound 3.5% DNBR reduction). Thus, it meets the DNB design basis and the results remain valid.

Rod Cluster Control Assembly Misalignment

An evaluation confirmed that the current state points (temperature, pressure and power) were applicable for uprated conditions. The DNB margin was also verified to accommodate the 1.4% uprating. The conclusions in the FSAR remain valid.

Partial and Complete Loss of Forced Reactor Coolant Flow

An analysis determined that the 1.4% uprating has a negligible effect on the transient state points. The current transient state points remain applicable and can be used with the increased nominal heat flux by 1.4% in the evaluation of DNB acceptance criteria. The analysis concluded that the DNB design basis continues to be met and the conclusions in the FSAR remain valid.

Loss of External Electrical Load and/or Turbine Trip

The licensing basis analysis for these events shows that there is a 38% analysis margin for Salem Unit No. 1 and 59% analysis margin for Salem Unit No. 2. This is sufficient safety analysis margin to offset the penalty associated with 1.4% uprating which amounts to 3.5% DNBR reduction. The results of the evaluation show that the DNB design basis continues to be met and the peak primary and secondary pressure remain below their respective limits. Thus, the FSAR conclusions remain valid.

Excessive Heat Removal Due to Feedwater System Malfunctions

PSEG performed the analyses under both full-power and no-load conditions to demonstrate that it would meet the DNB design basis. The analyses consider single-loop and multi-loop feedwater malfunctions and operation with both manual and automatic rod control. In the analyses, several variations were considered and the most limiting case shows over 8% margin to the safety analysis limit. Thus, there is a sufficient margin to offset the penalty associated with a 1.4% uprating, which is about 3.5% DNBR reduction.

Accidental Depressurization of the Reactor Coolant System

The current licensing basis analysis indicates that there is 29% safety analysis margin. There is a sufficient safety analysis margin to offset the penalty associated with a 1.4% uprating which is about 3.5% DNBR reduction. Thus, it meets the DNB design basis and the FSAR results remain valid.

Inadvertent Operation of Emergency Core Cooling System

This analysis assumes that the safety injection system is inadvertently actuated. One case assumes no direct reactor trip as a result of ECCS actuation. This case is inherently non-limiting as the DNBR increases through the duration of the transient. The minimum DNBR never falls below its initial value. Thus, it meets the DNB design criteria and the FSAR results remain valid. The other case is analyzed for pressurizer filling due to continued ECCS injection and reactor coolant expansion resulting from residual heat generation. This case assumes a reactor trip coincident with the event. Since this analysis is performed at 102% power, it bounds the proposed 1.4% uprate conditions in which power level uncertainty is limited to 0.6%.

Single Reactor Coolant Pump Locked Rotor

Two cases are considered. The first case is done to determine the percentage of fuel rods expected to experience DNB. The second case investigates the peak primary and secondary pressure transients with respect to RCS and main steam system pressure limits. In the first case, the initial power level is defined as the nominal full-power rating. The power level state points generated are in the form of fraction of the initial power. The 1.4% uprating has a negligible effect on the transient state points. As such, the current transient state points remain applicable and can be used with the nominal flux increased by 1.4%. The pressure case is analyzed with a 2% uncertainty included in the initial power level. Thus, the current analyses remain bounding for the 1.4% uprating and its associated 0.6% power level uncertainty, and FSAR conclusions remain valid.

Excessive Load Increase Accident

The analysis includes four different cases: Minimum and Maximum reactivity feedback with and without automatic rod control. The most limiting case is the minimum feedback/automatic rod control case. This case shows 38% safety analysis margin to the DNBR limit value. This will offset the 1.4% uprating penalty of 3.5%. Thus, it meets the DNB design basis and the FSAR results remain valid.

3.6.6.2 Non-LOCA Transient Analyses with Non-Statistical Methods

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition

PSEG analyzed this event at zero power; therefore the initial conditions are unaffected by the 1.4% uprating. The state points which are based upon a fraction of nominal conditions are unaffected by the 1.4% uprating. The limiting state points were evaluated with the increased heat flux associated with the 1.4% uprating. The evaluation shows that the DNB design basis is satisfied and the FSAR conclusions remain valid.

Rupture of Control Rod Drive Mechanism Housing

The current licensing basis analyses show 6% margin to the fuel enthalpy limit, 10% margin to the fuel melt limit, 49% margin to the Zircalloy clad metal-water reaction limit, and 67 °F margin to the peak cladding temperature limit. These margins are sufficient to offset any penalties associated with the small delay that could occur on the reactor trip time. Thus, the FSAR conclusions remain valid.

Accident Depressurization of the Main Steam System and Major Rupture of a Main Steamline

The safety analyses are performed under zero-power initial conditions to determine whether or not the DNBR limit is exceeded as a result of any potential recriticality. The transient state points remain unaffected by the 1.4% uprating. A detailed DNB evaluation with the increased nominal heat flux associated with the 1.4% uprating concluded that the DNB design basis continues to be met and the FSAR conclusions remain valid. The results of the major rupture of a main steamline analysis bound the results of the accidental depressurization analysis.

Uncontrolled Boron Dilution

This analysis ensures that there is sufficient time for operator action before loss of shutdown margin occurs. The critical parameters in the determination of the time available to terminate the dilution include the overall RCS active volume, the dilution flow rate, and the initial and critical boron concentrations. The licensing basis analysis assumes a conservative trip time of 120 seconds when only 89 seconds is needed. The existing analysis remains conservative and bounding. Thus, the FSAR conclusions remain valid.

3.6.7 Station Blackout

The Salem plants have been analyzed to a 4-hour station blackout (SBO) coping duration. For the proposed 1.4% power uprate it was confirmed that there is adequate AFW storage tank

inventory for the 4-hour coping duration. Therefore, the 4-hour coping duration for SBO remains valid for the proposed 1.4% power uprate.

3.6.8 Anticipated Transients Without Scram (ATWS)

Salem currently relies upon the generic Anticipated Transients Without Scram (ATWS) analyses performed by Westinghouse. The proposed power uprate to 3,459 MWt with a 0.6% uncertainty is bounded by the generic analyses.

3.7 Radiological Analysis of Design Basis Accidents

PSEG stated in Section 10.1 of its November 10, 2001, application, that the radiological consequence assessments for the following design basis accidents in the Salem FSAR have been performed at a reactor core thermal power level of 3,600 MWt, which is 105.5% of the current power rating of 3,411 MWt, bounding the proposed 1.4% power uprate:

- Loss-of-coolant accident
- Fuel handling accident
- Control rod rejection accident
- Locked rotor accident

PSEG further stated that the current radiological consequence assessments in the Salem Unit Nos. 1 and 2 FSAR remain valid for increased core thermal power operation. The staff reviewed the Salem FSAR and PSEG's amendment request describing the proposed increase in rated core thermal power. The staff's review of the radiological consequences analyzed for the above design basis accidents in Chapter 15 of the Salem FSAR confirmed that the current analyzed power level of 3,600 MWt bounds the requested uprate power level of 3,459 MWt, and that the radiological consequences calculated at a reactor core thermal power level of 3,600 MWt met the relevant dose acceptance criteria. Since the reactor accident source terms and release rates used in the current analyses remain bounding, the current calculated radiological consequences in the Salem FSAR remain bounding. PSEG proposed no change in methodology or assumptions for the radiological consequence assessments in its submittal.

In a license amendment request dated June 10, 1996, as supplemented June 24, July 1, August 13, September 30, and October 17, 1996, concerning radiation monitoring instrumentation and control room emergency air conditioning system, PSEG also analyzed the radiological consequences for a LOCA, fuel handling accident, locked rotor accident, and steam generator tube rupture accident at a reactor core thermal power level of 3,600 MWt and concluded that it met the relevant radiation dose reference values in 10 CFR Part 100 and the standard in GDC 19. In its review of the above license amendment request, the NRC staff performed independent radiological consequence analyses for these design basis accidents and concluded that PSEG's analysis supporting Salem License Amendment Nos. 190 (Unit No. 1) and 173 (Unit No. 2) was acceptable.

For the main steamline break accident outside containment, PSEG stated in Section 6.2 of the submittal that it evaluated steamline break mass and energy releases to determine the effect of a power uprate of up to 1.4% and determined that the NSSS design parameters, as described in Section 2, remain unchanged or bounded by the safety analysis values. PSEG further

concluded, and the staff agrees, that the main steamline break outside containment and instrument line pipe break scenarios are based on the design basis source terms of 1% failed fuel for noble gases and pre-accident and accident-initiated spikes for iodine; therefore, they are not impacted by the proposed 1.4% power uprate.

The staff has reviewed PSEG's amendment request and has concluded that the current design basis dose analyses, as documented in the Salem FSAR, remain acceptable in that reasonable assurance exists that the radiological consequences, with a proposed 1.4% reactor core thermal power uprate, will remain the same or are bounded by the current values. Therefore, the staff concludes that the proposed power uprate is acceptable with respect to the radiological consequences of the design basis accidents.

3.8 FOL and TS Changes

3.8.1 Changes to FOLs and TSs Revising Allowable Power Levels

The following changes to the Salem FOLs and TSs are required to reflect the new authorized power levels:

- Paragraph 2.C.(1) - FOLs DPR-70 and DPR-75 are revised to authorize operation at a steady state reactor core power level not to exceed 3,459 MWt.
- TS 1.25 - The definition of Rated Thermal Power is revised to reflect the increase from 3,411 MWt to 3,459 MWt.
- TS Figure 2.1-1 - Reactor Core Safety Limit is revised to reflect the new safety limits required to prevent core exit boiling at the new core power of 3,459 MWt.
- TS Table 2.2-1 - The Over-Temperature Delta Temperature F delta I penalties in TS Table 2.2-1, Reactor Trip System Instrumentation setpoints are revised to support the increase in core power.
- TS Table 3.7-1 - The maximum allowable thermal power with inoperable steamline safety valves and its bases are revised to reflect the increase in core power.
- TS 6.9.1.9 - The core operating limits report (COLR) is revised to add a reference to Topical Report CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," dated May 2000.

The FOL and TS changes reflect the actual proposed changes in the plant and they are consistent with the results of the safety analysis. Accordingly, the NRC staff finds these changes acceptable.

3.8.2 TS Section 3/4.4.9 Pressure/Temperature (P-T) Limits

3.8.2.1 Introduction

Regarding issues associated with RPV integrity, PSEG's submittal included a request to revise each unit's TS Figures 3.4-2 and 3.4-3 to implement new RPV pressure-temperature (P-T) limit curves (for heatup, cooldown, core criticality, and hydrostatic/leak rate testing) valid through 32 EFPY of operation and a pressurized thermal shock (PTS) reassessment. The development of the revised P-T limit curves was based on an associated request for an exemption to utilize ASME Code Case N-640. By letter dated December 5, 2000, PSEG supplemented their original submittal. The supplemental submittal forwarded reports WCAP-15565, Revision 0 and WCAP-15566, Revision 0 regarding the development of the requested P-T limit curves. Finally, PSEG responded by letter dated March 28, 2001, to an NRC staff RAI, and, as part of the response, submitted Revision 1 of each of the aforementioned reports for staff review along with a revised TS amendment request.

3.8.2.2 Background

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. These requirements include:

(1) Appendix G to 10 CFR Part 50 which requires the establishment of P-T limit curves, and the evaluation of Charpy V-notch upper shelf energy drop; and (2) 10 CFR 50.61 which establishes requirements for protecting RPVs from potential failure as a result of PTS events.

The staff evaluated the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; GL 88-11; GL 92-01, Revision (Rev.) 1; GL 92-01, Rev. 1, Supplement 1; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the decrease in reactor vessel material fracture toughness (as quantified by the increase in transition temperature and the decrease in upper-shelf energy (USE)) resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their RPV data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the review of P-T limit curves. Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the Code.

SRP Section 5.3.2 provides an acceptable method for determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the Code. The basic parameter of this methodology is the stress intensity factor K_I , which is a function of the stress state and flaw configuration. Appendix G to Section XI requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 on the same stresses for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the 1/4 thickness

(1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The methodology of ASME Code, Section XI, Appendix G, requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT} , where RT_{NDT} is the reference temperature at the nil ductility transition for the material). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term by application of RG 1.99, Revision 2. The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor (CF) was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT} , the copper and nickel contents, the fluence and the calculational procedures. RG 1.99, Revision 2, describes the methodology to be used in calculating the margin term.

The PTS rule, 10 CFR 50.61, requires that licensees demonstrate that facility RPV materials will continue to possess an adequate level of fracture resistance to protect the RPV from potential failure as a result of PTS events. Each material's PTS reference temperature, RT_{PTS} , is determined in a manner like that used to determine ART, except that the neutron fluence at the clad-to-base metal interface at end of license (EOL) conditions is used in lieu of either the 1/4T or 3/4T fluence. Each material's RT_{PTS} value is then compared to the screening limits given in 10 CFR 50.61, which are 270 °F for plates, forging, and axial welds, and 300 °F for circumferential welds. Provided that all RPV materials' RT_{PTS} values remain below these screening limits, the fracture resistance of the RPV is demonstrated to be adequate to meet the requirements of 10 CFR 50.61 through EOL.

3.8.2.3 Evaluation

Licensee's Evaluation

PSEG requested, pursuant to 10 CFR 50.60(b), an exemption to use ASME Code Case N-640 as the basis for establishing the P-T limit curves. Code Case N-640 permits application of the lower bound static initiation fracture toughness value equation (K_{Ic} equation) as the basis for establishing the P-T curves in lieu of using the lower bound crack arrest fracture toughness value equation (i.e., the K_{Ia} equation, which is based on conditions needed to arrest a dynamically propagating crack, and which is the method invoked by Appendix G to Section XI of the ASME Code).

As amended by information included with its March 28, 2001, letter, PSEG submitted beltline material ART calculations and P-T limit curves valid through 32 EFPY for Salem Unit Nos. 1 and 2. For the Salem Unit No. 1 RPV, PSEG determined that the limiting beltline material at the 1/4T through-wall location was the lower shell axial weld 3-042C, and that the limiting beltline material at the 3/4T through-wall location was intermediate shell plate B-2402-1. For the Salem Unit No. 2 RPV, PSEG determined that the limiting beltline material at both the 1/4T and 3/4T throughwall locations were lower shell axial welds 3-442 A and C (both of these welds are fabricated from the same heat of weld wire and receive the same neutron fluence; hence, both

are projected to have identical material properties). Tables 1 and 2 summarize the relevant information used to establish the 32 EFPY ART for the limiting beltline materials for each unit at each throughwall location.

For convenience, the final row in each table summarizes the information submitted by PSEG and used to determine the RT_{PTS} value for each RPV's limiting beltline material based on PTS considerations. Based on this information, PSEG concluded that the material properties of Salem Unit Nos. 1 and 2 vessels would continue to meet the requirements of 10 CFR 50.61 through EOL.

Table 1: Determination of Limiting Beltline Material ARTs for Salem Unit 1 @ 32 EFPY

Material	Cu/Ni/CF	Fluence	IRT_{NDT}	ΔRT_{NDT}	Margin	ART/ RT_{PTS}
Lower Shell Axial Weld 3-042C	0.19/1.04/223.6	9.77×10^{18} (b)	-56	222	65.5	232
Intermediate Shell Plate B-2402-1	0.24/0.53/153.6 ^(a)	3.47×10^{18} (c)	45	109	17	171
Lower Shell Axial Weld 3-042C	0.19/1.04/223.6	1.64×10^{19} (d)	-56	254	65.5	264

^(a) CF value determined from credible surveillance data

^(b) 1/4T fluence value

^(c) 3/4T fluence value

^(d) Clad-to-base metal interface fluence value

Table 2: Determination of Limiting Beltline Material ARTs for Salem Unit 2 @ 32 EFPY

Material	Cu/Ni/CF	Fluence	IRT_{NDT}	ΔRT_{NDT}	Margin	ART/ RT_{PTS}
Lower Shell Axial Welds 3-442 A&C	0.213/0.867/208.6	7.15×10^{18} (b)	-56	189	65.5	199
Lower Shell Axial Welds 3-442 A&C	0.213/0.867/208.6	2.54×10^{18} (c)	-56	131	65.5	140
Lower Shell Axial Welds 3-442 A&C	0.213/0.867/208.6	1.20×10^{19} (d)	-56	219	65.5	229

^(b) 1/4T fluence value

^(c) 3/4T fluence value

^(d) Clad-to-base metal interface fluence value

NRC Staff's Evaluation

As previously mentioned, PSEG requested an exemption to use ASME Code Case N-640 as the basis for establishing the P-T limit curves. The K_{Ic} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The staff concluded that P-T curves based on the K_{Ic} curve will enhance overall plant safety by opening the P-T operating window with the greatest safety benefit in the region of low temperature operation. Approval of PSEG's exemption request was granted on May 25, 2001.

The staff performed an independent calculation of the ART values for the limiting beltline materials using the methodology in RG 1.99, Revision 2. Based on these calculations, the staff verified that PSEG identified the appropriate limiting beltline materials for both Salem Unit Nos. 1 and 2. The staff's calculated 1/4T ART, 3/4T ART, and RT_{PTS} values for the limiting beltline materials agree with the values calculated by PSEG.

In addition to the beltline material properties previously discussed, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Because the specified testing is performed at greater than 295 °F and normal operating temperature is greater than 530 °F, the NRC staff concludes that the limiting flange RT_{NDT} values of 60 °F for Salem Unit No. 1 and 28 °F for Salem Unit No. 2 are acceptable.

The staff evaluated PSEG's P-T limit curves for acceptability by performing independent calculations, using the methodology referenced in the ASME Code (as indicated by SRP 5.3.2) and as modified by the use of ASME Code Case N-640. The staff determined that the initial set of revised P-T limit curves submitted by PSEG in their November 10, 2000, letter were not acceptable. PSEG revised their proposed P-T limit curves and resubmitted them in response to an NRC staff RAI by letter dated March 28, 2001. Based on its independent calculations, the staff concludes that the P-T limit curves provided with PSEG's March 28, 2001, letter meets the requirements of Appendix G of Section XI of the ASME Code, as modified by Code Case N-640. Therefore, the staff has determined that PSEG's proposed P-T limit curves for heatup, cooldown, core criticality, and hydrostatic/leak rate testing are acceptable, since they meet the requirements of 10 CFR 50.60 and Appendix G of 10 CFR Part 50.

Finally, the staff reviewed the revised PTS assessment for Salem Unit Nos. 1 and 2. The staff concludes that the information submitted by PSEG meets the requirements of 10 CFR 50.61, and is acceptable since it demonstrates that the limiting materials for both Salem Unit Nos. 1 and 2 would not exceed the PTS screening criteria through EOL. Therefore, the staff finds that both units will remain in compliance with the requirements of 10 CFR 50.61 through EOL. The information used to support this finding will be incorporated into the NRC staff's Reactor Vessel Integrity Database.

3.8.2.4 P-T Limit Curves Evaluation Conclusions

The NRC staff concludes that the P-T limits curves proposed in PSEG's March 28, 2001, letter for heatup, cooldown, core criticality, and hydrostatic/leak rate testing satisfy the requirements in Appendix G to Section XI of the ASME Code, as modified by Code Case N-640, and Appendix G of 10 CFR Part 50. The proposed P-T limits also satisfy GL 88-11, because the method in RG 1.99, Revision 2, was used to calculate the beltline material ARTs. Hence, the proposed P-T limit curves may be incorporated into the Salem Unit Nos. 1 and 2 TSs. The staff also concludes that the PTS assessment has demonstrated that both units will remain in compliance with the requirements of 10 CFR 50.61 through EOL.

3.8.3 Editorial Changes and Removal of Historical Information

The proposed amendment includes changes to the TS Bases to correct references and typographical errors. The staff considers these changes to be editorial in nature and notes that they are consistent with the proposed license amendment. In addition, PSEG proposed to remove references to preoperational tests, startup tests and other items associated with the initial startup of Salem Unit No. 1 identified in Paragraph 2.C.(1) of the FOL. The staff concurs that this information is no longer necessary and finds that the changes are acceptable.

3.8.4 Bases to the TSs

PSEG submitted changes to the Bases for the specifications proposed to be changed in the license amendments. Primarily, the Bases of the TSs are incorporating new references to topical reports, revising values associated with RT_{NDT} and P-T limit curves, and changing other sections related to the amended TSs. The NRC staff notes that the Bases changes proposed for the TSs are consistent with the proposed license amendment.

3.9 Human Factors

3.9.1 Changes in Emergency and Abnormal Operating Procedures

PSEG stated in its letter dated November 10, 2000, that, "[p]lant procedures will not require significant changes for the uprate." Procedural limitations on power operation, because of certain balance of plant (BOP) equipment being unavailable, will be revised as needed "to account for the increase in NSSS power to 3,471 MWt. The only new procedures required are for operation and maintenance of the Crossflow system."

The staff finds that PSEG's response is satisfactory because, consistent with RG 1.33, PSEG has adequately identified the type and scope of plant procedures that will be affected by the uprate, indicated that the procedures will be appropriately revised, operators will be trained on the changes before the procedures are implemented, and adequately described the effect of the procedure changes on operator actions.

3.9.2 Changes to Risk-Important Operator Actions Sensitive to Power Uprate

PSEG stated in its letter dated November 10, 2000, that, "the ESF System design and setpoints, and procedural requirements [sic] are already bound the proposed uprate. The

uprate will not change the time available for the operators to respond, or add additional steps.” PSEG further stated that, though the uprate will “reduce the margin available during the limiting BOP transients, it does not change the required operator response.” PSEG indicated that even the most limiting steam generator feed pump (SGFP) trip transient “does not impose any new requirements for operator response.”

The staff finds that PSEG’s response is satisfactory because it has adequately addressed the question of operator actions sensitive to the power uprate by describing the effect of the power uprate on operator performance and adequately justifying the effect/lack of effect on required operator response.

3.9.3 Changes to Control Room Controls, Displays and Alarms

In its November 10, 2000, letter, PSEG stated that, “there will be minimum impact on alarms, controls, and displays for a 1.4% uprate.” Indicated power (i.e., reactor power 100%) will be re-scaled to indicate the new uprated power. “Therefore, the increased megawatt rating will indicate at 100% power.” Installation of the Crossflow UFM systems used for feedwater flow measurements will result in the subsequent installation of additional alarms in the main control room to alert the operators to conditions of UFM inoperability or inaccuracy. “No other alarm impacts are expected.” PSEG stated that alarms will be re-calibrated as required to reflect setpoint changes though “no significant or fundamental setpoint changes are anticipated.”

The staff finds that PSEG’s response is satisfactory because it has adequately identified the changes that will occur to alarms, displays, and controls as a result of the power uprate and adequately described how these changes will be accommodated.

3.9.4 Changes on the Safety Parameter Display System

PSEG indicated that, “[p]rocess parameter scaling changes will be made as required for the Safety Parameter Display System (SPDS). There are no other impacts to the SPDS from the proposed uprate. Implementation of scaling changes made will be controlled under PSEG Nuclear’s software configuration change control program.”

The staff finds that PSEG’s response is satisfactory because it has adequately identified the changes that will occur to the SPDS as a result of the power uprate and adequately described how the changes will be addressed.

3.9.5 Changes to the Operator Training Program and the Control Room Simulator

PSEG indicated that, “[s]ince the power uprate is nominal and there is no change to how the plant will be operated, the impact on operator training is minimal.” PSEG also indicated that “[t]he effect on the plant simulator will be minimal. The simulator initial conditions will be revised to account for the increase from 3411 to 3459 MWt as 100% power. The simulator OTΔT neutron flux penalties will be revised to reflect the revised flux penalties described in the proposed changes to the Technical Specifications.” PSEG will also add an overhead annunciator window to alert operators to Crossflow trouble. No other changes to the simulator were identified.

The staff finds PSEG's response satisfactory because it has adequately described how the changes to operator actions will be addressed by the simulator and how the simulator will accommodate the changes in accordance with the requirements of ANS/ANSI Standard 3.5.

3.9.6 Human Factors Evaluation Conclusion

The staff concludes that the previously discussed review topics associated with the proposed power uprate have been satisfactorily addressed. The staff further concludes that the power uprate should not adversely affect simulation facility fidelity, operator performance, or operator reliability.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the *Federal Register* on May 15, 2001 (66 FR 26885). Accordingly, based upon the Environmental Assessment, the staff has determined that issuance of the amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from D. F. Garchow, PSEG, to USNRC, "Request for License Amendment Increased Licensed Power Level Salem Generating Station, Units 1 and 2 Facility Operating License DPR-70 and DPR-75 Docket Nos. 50-272 and 50-311," dated November 10, 2000.
2. Letter from Stuart A. Richards (NRC) to Ian C. Rickard, PSEG, "Acceptance for Referencing of CENPD-397-P, Revision-01-P," dated March 20, 2000.
3. CENPD-397-P-A, Revision 1, "Improved Flow Measurement Accuracy Using Cross-Flow Ultrasonic Flow Measurement Technology," Combustion Engineering, dated May, 2000.
4. Letter from D. F. Garchow, PSEG, to USNRC, "Supplement to Request for License Amendment Increased Licensed Power Level Salem Generating Station, Unit Nos. 1 and 2, Facility Operating License DPR-70 and DPR-75, Docket Nos. 50-272 and 50-311," dated December 5, 2000.

5. Letter from D. F. Garchow, PSEG, to USNRC, "Response to Request for Additional Information in Regards to Request for License Amendment Increased Licensed Power Level, Salem Generating Station, Unit Nos. 1 and 2, Facility Operating License DPR-70 and DPR-75 Docket Nos. 50-272 and 50-311," dated March 28, 2001.
6. Letter from M. B. Bezilla, PSEG, to USNRC, "Supplemental Environmental Information in Regards to Request for License Amendment Increased Licensed Power Level, Salem Generating Station, Unit Nos. 1 and 2, Facility Operating License DPR-70 and DPR-75, Docket Nos. 50-272 and 50-311," dated April 2, 2001.
7. Letter from G. Salamon, PSEG, to U.S. NRC "Response to March 16, 2001, Request for Additional Information in Regards to Request for License Amendment Increased Licensed Power Level, Salem Generating Station, Units 1 and 2, Facility Operating License DPR-70 and DPR 75, Docket Nos. 50-271 and 50-311," dated April 20, 2001.
8. Letter from G. Salamon, PSEG, to U.S. NRC "Response to April 3, 2001, Request for Additional Information in Regards to Request for License Amendment Increased Licensed Power Level, Salem Generating Station, Units 1 and 2, Facility Operating License DPR-70 and DPR 75, Docket Nos. 50-271 and 50-311," dated April 20, 2001.
9. Letter from G. Salamon, PSEG, to U.S. NRC "Response to April 12, 2001, Request for Additional Information in Regards to Request for License Amendment Increased Licensed Power Level, Salem Generating Station, Units 1 and 2, Facility Operating License DPR-70 and DPR 75, Docket Nos. 50-271 and 50-311," dated April 20, 2001.

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Date: May 25, 2001

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